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March 11, 1998

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: McGuire Nuclear Station, Unit 1
Docket No. 50-369
Licensee Event Report, 369/98-02
Problem Investigation Process No.: 1-M98-0386

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 369/98-02 concerning a reactor trip at McGuire Unit 1. This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv)(A). This event is considered to be of no significance with respect to the health and safety of the public.

This LER contains regulatory commitments in the planned corrective actions section.

Very truly yours,

H B Barron

H. B. Barron, Jr.

Attachment

cc: L. A. Reyes
U.S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
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Atlanta, GA 30323

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Washington, D.C. 20555

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NRC Resident Inspector
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1141

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
McGuire Nuclear Station 1	05000 369	1 of 7

TITLE (4)
McGuire 1 Manual Reactor Trip following a fuse failure in the rod control system

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S)
										05000
02	09	98	1998	- 369 - 02 -	0	03	11	98	McGuire Nuclear Station 1	05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (*Check one or more of the following*) (11)							
POWER LEVEL (10)	100%		20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
			20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (*Specify in Abstract below and in Text, NRC Form 366A*)
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
M. T. Cash	(704)	875-4117

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B2	JD	EIIS:FUB	Littelfuse	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (*if yes, complete EXPECTED SUBMISSION DATE*)	X	NO		n/A	n/a	n/a

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

Unit Status: Unit 1 Mode 1 at 100% Reactor Power

Event Description:
Intermittent operation of a fuse holder [EIIS:FUB] caused 8 control rods to drop into the core and 3 rods in control bank D to move in approximately 50 steps. Operators manually tripped the reactor prior to reaching an automatic reactor trip setpoint.

Event Cause:
A power supply failure occurred due to an intermittent loss of power to the rod control [EIIS: JD] gripper mechanism. The momentary loss of AC power to the functioning power supply was caused by intermittent operation of a fuse holder.

Corrective Action:
The fuse holder was replaced and other fuse holders were inspected and replaced as appropriate. Additional procedural guidance will be provided for replacing defective rod control power supplies on-line.

The existing shutdown procedure on rod control power supply will be enhanced to check the integrity of the fuse holders.

NRC FORM 366*NPRDS no longer exists, the failure will be reported through EPIX

EVALUATION:

Description of Event

Unit was in MODE 1 (Power Operation) at 100% Power

Initial Rod Control Alarm and Trouble Shooting
Unit 1 was at 100% power when a non-urgent rod control alarm was received on power cabinet 2BD of the rod control system.

Instrument and control technicians determined that PS3 (power supply 3) had low voltage. Operators placed control rods in manual.

A replacement power supply was prepared by bench testing and wiring verification. The new power supply was installed in the rod control [EIIS: JD] cabinet. The fuses [EIIS: FU] for this supply, FU3A and FU3B, were removed for the installation. The field wiring was terminated on the new power supply terminal strip. The next step required installation of the fuses and powering up the power supply [EIIS: RJX] to clear the non-urgent alarm. When fuse FU3B (the first of two supply fuses, AC common line) was installed, the operators received an URGENT rod control alarm. It was initially suspected that the new power supply had caused the URGENT alarm. Fuse FU3B was removed and work was stopped to determine the problem.

Power supply wiring to the external terminal strip was verified prior to resuming work. The field wiring to PS3 was removed and fuses FU3A and FU3B were inserted. The output of the supply was verified correct. The field wiring was checked against drawings to ensure proper connection and polarity. It was concluded that the URGENT alarm was not associated with the power supply failure, and a decision was made to continue with power supply replacement before troubleshooting the cause of the urgent alarm.

Work resumed by pulling fuses FU3A and FU3B. The field wiring was terminated to PS3. Fuse FU3B was reinstalled first with no adverse condition noted by the technicians. Fuse FU3A was then installed, and immediately the light on FU4B illuminated, indicating a blown fuse or open circuit. The momentary loss of both PS3 and PS4 resulted in dropping group 2 of Control Bank B and group 2 of Shut Down Bank B fully into the core. Control Bank D group 2 rods dropped from approximately 210 steps to 156 steps. Voltage readings were taken confirming a loss of output voltage from PS4. The technician touched FU4B, which caused the fuse holder light to extinguish. The output voltages of both PS3 and PS4 were checked and found to be appropriate.

Plant response to dropped Rods
Operators entered AP/1/A/5500/14 "Rod Control Malfunction" and noted dropping steam pressure, pressurizer level and pressurizer pressure as a result of the large negative reactivity insertion associated with the dropped control rods.

Nuclear Power as indicated by the excore power range detectors indicates that power level instantaneously dropped to approximately 50%. Within seconds the power level recovered and stabilized at approximately 70% power due to moderator temperature feedback effects. The rate of decrease of turbine generator load was not adequate to compensate for the primary system cooldown associated with the negative reactivity insertion.

Therefore, operators made a decision to manually trip the reactor approximately one minute after the rods had moved into the core and prior to reaching any automatic reactor trip setpoint.

The turbine tripped as designed due to the reactor trip.

Approximately five seconds after the reactor trip there was an automatic auxiliary feedwater [EIIS: BA] start and a main feedwater isolation. The feedwater isolation occurred as a result of LO LO T_{avg} coincident with a reactor trip. Main feedwater [EIIS: SJ] was placed back in service approximately two hours later and auxiliary feedwater removed from service. Pressurizer level decreased in response to the plant cooldown, reaching the low level setpoint for letdown isolation approximately six seconds following the reactor trip. Letdown was placed back in service approximately 11 minutes later.

The AUTOSTART of Auxiliary Feedwater was an expected response due to AMSAC logic. AMSAC logic is enabled when power exceeds 40%. AMSAC logic is disabled, with a 120 second delay, following a power decrease below 40%. AMSAC will initiate the AUTOSTART of Auxiliary Feedwater on the closure of Main Feedwater Isolation or Main Feedwater Regulating valves. The Main Feedwater Isolation satisfied the AMSAC logic within the 120 second delay following the drop below 40% power. Auxiliary Feedwater received a valid AUTOSTART signal.

The four channels of nuclear instrumentation [EIIS: IG] indicated differing responses during the rod drop. This was due primarily to the location of the various dropped and mis-aligned Control Bank D rods with respect to the individual channels of nuclear instrumentation.

At the time of the rod drop, all four power range detectors dropped from an initial indication of approximately 100% to approximately 46% power.

Immediately following the rod drop, the slope of the power recovery for N-43 and N-44 exceeded that of N-41 and N-42 for approximately the first five seconds. Following this pronounced change in slope, all power range detectors responded in a similar manner up to and following the manual reactor trip.

Summary Sequence of Events

01:45 Rod Control Non-Urgent Failure Alarm Received.

02:25 IAE determined power supply 3 in power cabinet 2BD has failure.

07:00 Entered AP/1/A/5500/14, Control Rod Malfunction

10:26:03 Power supply failure in cabinet 2BD during power supply swap. Rods Shutdown Bank B, Group 2 Control Bank B, Group 2 (8 rods total) fall into the core. Control Bank D, Group 2 (3 rods) dropped approximately 50 steps to approximately 150 steps.

Decreasing main steam pressure, pressurizer pressure, pressurizer level.

10:27:09 Manual reactor trip initiated due to falling steam pressure, pressurizer pressure and pressurizer level.

10:27:09 Turbine Trip.

10:27:10 Pressurizer Low Pressure Reactor Trip setpoint reached.

10:27:14 Main Feedwater isolation on Reactor Trip coincident with LO-LO T_{avg}

10:27:14 Auxiliary Feedwater auto start on AMSAC.

10:27:15 Letdown isolation on low pressurizer level.

10:38:46 Pressurizer level recovered and letdown placed back in service.

12:18:34 Main feedwater re-established, auxiliary feedwater isolated.

Conclusion

This event did not result in any uncontrolled releases of radioactive material, personnel injuries or radiation overexposure. This event is EPIX reportable.

Cause Analysis

The failure was determined to be an intermittent fuse holder (Littelfuse 344125). This failure was re-created in the field subsequent to the event. The intermittent failure was observed while replacing a fuse in a holder adjacent to the suspect holder. The holder illuminated briefly, suggesting that the fuse itself had blown; however, the fuse (3AG) had not blown. This is consistent with the observations of the technicians and the engineer during the actual event.

The fuse holder was originally supplied by Westinghouse in the rod control power cabinet. The fuse holder is estimated to be a 25 year old component. The cap was found to be loose prior to removal, even though it was installed properly in the 12 o'clock position.

The suspect fuse holder was replaced with a new one. The suspect holder was subjected to electrical and metallurgical laboratory testing and analysis. In summary, the test results were inconclusive. Visual inspection under magnification revealed a number of imperfections; however, the failure could not be repeated in the lab. In addition, electrical testing revealed no failures except when the holder's cap was oriented other than the latched, 12 o'clock position. This was not the as-found position of the cap. A metallurgical analysis was conducted to determine if oxidation, debris. other contamination, or mechanical deformation could have contributed to the failure. Other areas of scrutiny included the spring tension, mechanical contact, and the fuse itself. The metallurgical report could not establish an absolute cause for the intermittent behavior of the fuse holder. The report also stated that non-conductive contamination was found inside the holder and imbedded in the bottom contact seating surface. It is possible this condition contributed to a momentary loss of continuity.

The cause of the initial URGENT alarm was concluded to be a brief circuit interruption from PS4 while PS3 was out of service. This interruption was not of sufficient duration to cause rod movement. The URGENT alarm remained in place throughout the event because it is required to be manually reset in order to clear.

CORRECTIVE ACTION:

Immediate

Operators placed rods in manual during rod control troubleshooting. Operators manually tripped the reactor and stabilized the plant following the reactor trip.

All fuse holders in the rod control system were examined following this event. Three suspect holders were replaced in addition to the failed holder. Unit 2 fuse holders have been inspected and found satisfactory.

Subsequent

Prior to reactor startup, the rod control system was tested to demonstrate normal operation.

Planned

Additional procedural guidance will be provided for replacement of defective rod control power supplies on-line.

The existing shutdown procedure for rod control power supply PM will be enhanced to provide a check of the integrity of these fuse holders and provide for any necessary replacements.

SAFETY ANALYSIS:

Based on the following analysis, this event is not considered to be significant. At no time was the safety or health of the public or plant personnel affected as a result of this event.

The licensing basis dropped rod event, UFSAR Section 15.4.3d, assumes that any combination of control rods from the same group drop into the core from full power. Beginning, middle, and end-of-cycle core conditions are analyzed to ensure that the limiting transient is identified and analyzed. The primary acceptance criterion which limits this event is the DNB criterion. The analysis is performed to ensure that the DNBR limit is not exceeded for any combination of dropped rods from the same group at any time in core life.

The primary parameters which affect DNBR are the core average power level, the core peaking factors, flow, pressure, and temperature. The licensing basis analysis assumes limiting values for all of these parameters, based on the full power initial conditions and the post-event transient response. The initial core peaking factor is described by a pin radial-local peaking factor of 1.50. The post-event peaking factors are similarly quantified by analysis, and reflect the more highly peaked core power distribution due to the dropped rod(s). For the licensing basis limiting case, the core average power level is greater than 100%, the pin radial-local peaking factor is

1.70, and the axial peaking factor is 1.38. The minimum DNBR for all licensing basis events is greater than the 1.40 limit.

For the McGuire Unit 1 event, the pre-event pin radial-local peaking factors was 1.42. The post-event peaking factors were a pin radial-local of 1.71 and an axial peak of 1.14. The core average power was approximately 73%. Although the pin radial factor slightly exceeded the licensing basis case value, the other DNB related parameters had significant margin as compared to the licensing basis case. It can be concluded that the McGuire 1 dropped rod event was well bounded by the UFSAR licensing basis DNBR and did not approach the 1.40 limit.

CATEGORY 1

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SUBJECT: Forwards LER 98-002-00,concerning reactor trip at McGuire Unit 1.Commitments made within ltr,encl.

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